TVA

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

September 7, 2007

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

In the Matter of	)	Docket No. 50-391
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT (WBN) - UNIT 2 - INITIAL RESPONSES TO BULLETINS AND GENERIC LETTERS

The purpose of this letter is to provide the initial responses for WBN Unit 2 for the following Bulletins and Generic Letters:

- Bulletin 96-01 - Control Rod Insertion Problems (PWR)
- Bulletin 96-02 - Movement of Heavy Loads
- Bulletin 01-01 - Cracking of RPV Head Penetration Nozzles
- Bulletin 02-01 - RPV Head Degradation and Reactor Coolant Pressure Boundary Integrity
- Bulletin 02-02 - RPV Head and Vessel Head Penetration Nozzle Inspection Program
- Bulletin 03-01 - Potential Impact of Debris Blockage on Emergency Sump Recirculation
- Bulletin 03-02 - Leakage from RPV Lower Head Penetrations & Reactor Coolant Pressure Boundary Integrity

A056 A109 A122
A072 A110 A123
A076 A103 A125
A079 A115 A127
A116 D030.

NRR

- Bulletin 04-01 - Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at PWRs
- Generic Letter 95-03 - Circumferential Cracking of Steam Generator (SG) Tubes
- Generic Letter 95-05 - Voltage Based Repair Criteria for W SG Tubes Affected by Outside Diameter Stress Corrosion Cracking
- Generic Letter 95-07 - Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves
- Generic Letter 96-06 - Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions
- Generic Letter 97-04 - Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps
- Generic Letter 97-05 - SG Tube Inspection Techniques
- Generic Letter 97-06 - Degradation of SG Internals
- Generic Letter 98-02 - Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition
- Generic Letter 98-04 - Potential for Degradation of the ECCS and the Containment Spray System After a LOCA Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment
- Generic Letter 03-01 - Control Room Habitability
- Generic Letter 04-01 - Requirements for SG Tube Inspection
- Generic Letter 04-02 - Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at PWRs
- Generic Letter 06-01 - SG Tube Integrity and Associated Technical Specifications
- Generic Letter 06-02 - Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power
- Generic Letter 06-03 - Potentially Nonconforming Hemyc and MT Fire Barrier Configurations
- Generic Letter 07-01 - Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients

These Bulletins and Generic Letters were issued to holders of operating licenses. Because Watts Bar Unit 2 was in a deferred construction status, TVA was not required to respond. In Reference 1, TVA informed the Nuclear Regulatory Commission (NRC) Staff of TVA's intention to reactivate and complete construction activities at WBN Unit 2. In preparation for requesting an operating license, TVA must demonstrate that WBN Unit 2 is in compliance with applicable regulations.

Attachments 1 through 22 to this letter provide the initial WBN Unit 2 response to the specific Bulletin or Generic Letter. Each attachment provides the appropriate references and, based on the WBN Unit 1 precedent, the actions TVA will take to resolve the issue. TVA's objective in this regard is to align the licensing and design bases of Watts Bar Units 1 and 2 to the fullest extent practicable. In summary, TVA intends to implement the WBN Unit 1 solution to the Bulletin or Generic Letter for WBN Unit 2. Attachment 23 provides a listing of the commitments made in this submittal. Implementation of the commitments will be provided under the WBN Unit 2 construction procedures until the appropriate turnover milestone.

If TVA determines based on discovery or emerging issues that a different strategy or additional action is appropriate, TVA will submit such changes to the NRC for review and concurrence. TVA will continue to review generic communications as the WBN Unit 2 Regulatory Framework is developed.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 7th day of September, 2007. If you have any questions, please contact me at (423) 365-2351.

Sincerely,

M. Bajestani

Masoud Bajestani
Watts Bar Unit 2 Vice President

References:

1. TVA letter dated August 3, 2007, William R. McCollum, Jr. to NRC, "Watts Bar Unit 2 - Reactivation of Construction Activities".

Attachment

(cc w/ Attachment):

Catherine Haney, Director
U.S. Nuclear Regulatory Commission
MS 08G9
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2738

Lakshminarasimh Raghavan
U.S. Nuclear Regulatory Commission
MS 08H4A
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2738

Loren R. Plisco, Deputy Regional Administrator for Construction
U. S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center, Suite 23T85
61 Forsyth Street, SW,
Atlanta, Georgia 30303-8931

U. S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

JEM:
Enclosure
cc (w/ Attachment):

G. Arent, EQB 1B-WBN
M. Bajestani, EQB 1B-WBN
R. R. Baron, EQB 1B-WBN
A. S. Bhatnagar, LP 6A-C
R. H. Bryan, BR 4X-C
J. C. Fornicola, LP 6A-C
M. D. Skaggs, ADM-1V-WBN
J. Valente, EQB 1B-WBN
E. J. Vigluicci, ET 11A-K
EDMS, WT 3B-K

Attachment 1

NRC BULLETIN 96-01: CONTROL ROD INSERTION PROBLEMS

Watts Bar Unit 2 will demonstrate operability of the rod control system as part of the Power Ascension Test Program by performance of the following tests:

- Refueling and Core Alterations (includes drag test)
- Control Rod Drive Mechanism Timing
- Rod Position Indication System
- Rod Drop Testing
- Rod Drop Time Measurements

The current Emergency Operating Instruction ES-0.1 has the Reactor Operator ensure all control rods are fully inserted as indicated by the rod position indication system. This procedure initiates boration if two or more control rods are not fully inserted. A similar procedure will be issued for Unit 2 prior to startup.

TVA will provide a core map of rodded fuel assemblies indicating fuel type (materials, grids, spacers, guide tube inner diameter) and projected end of cycle burnup of each rodded assembly for the initial fuel cycle. This information will be provided six months prior to fuel load.

Attachment 2

NRC BULLETIN 96-02: MOVEMENT OF HEAVY LOADS OVER SPENT FUEL, OVER FUEL IN THE REACTOR CORE, OR OVER SAFETY-RELATED EQUIPMENT

Requested Actions:

To ensure that the handling of heavy loads is performed safely and within the conditions and requirements specified under Title 10 of the Code of Federal Regulations, all addressees are requested to take the following actions:

Review plans and capabilities for handling heavy loads while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) in accordance with existing regulatory guidelines. Determine whether the activities are within the licensing basis and, if necessary, submit a license amendment request. Determine whether changes to Technical Specifications will be required in order to allow the handling of heavy loads (e.g., the dry storage canister shield plug and associated lifting devices) over fuel assemblies in the spent fuel pool.

TVA Response: Watts Bar Nuclear Plant Unit 1 and Unit 2 have two common storage areas, one for new fuel and one for spent fuel. Heavy load lifts over fuel assemblies are performed under the operating license for Unit 1. The WBN Technical Requirements Manual (TRM) prohibits loads greater than 2059 pounds from travel over fuel assemblies. This ensures that objects traversing the pool are within the design basis and will not cause an unsafe condition if accidentally dropped.

As part of TVA's response to NUREG-0612 (Reference 1), TVA committed that the Watts Bar Unit 2 Heavy Loads Program would be in compliance with requirements by Unit 2 fuel load.

References:

1. TVA letter dated July 28, 1993, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 – Generic Letter (GL) 81-07 – NUREG-0612 – Control of Heavy Loads at Nuclear Power Plants – Revised Response – License Condition (LC) 39 – (TAC NOS. M77560 and M77561).

Attachment 3

NRC BULLETIN 2001-01: CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES

NRC BULLETIN 2002-01: REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

NRC BULLETIN 2002-02: REACTOR PRESSURE VESSEL HEAD AND VESSEL HEAD PENETRATION NOZZLE INSPECTION PROGRAMS

To meet the requirements of Bulletins 2001-01, 2002-01 and 2002-02, Watts Bar Unit 2 will implement the inspection and reporting requirements for a plant in the low category of Reference 1. Specifically, Watts Bar Unit 2 will perform the first inspections meeting the requirements of paragraphs IV.C(5)(a) and IV.C(5)(b) of Reference 1 at the first refueling outage.

TVA will perform a baseline inspection prior to fuel load.

References:

1. NRC letter dated February 20, 2004 to Holders of Licenses for Operating Pressurized Water Reactors, "Issuance of First Revised NRC Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors"

Attachment 4

NRC BULLETIN 2003-01: POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY SUMP RECIRCULATION AT PRESSURIZED-WATER REACTORS

Bulletin 2003-01 requests that TVA describe any interim compensatory measures that have been implemented or that will be implemented to reduce the risk which may be associated with potentially degraded or nonconforming ECCS and CSS recirculation functions until an evaluation to determine compliance is complete. As discussed in Attachment 18 for Generic Letter 2004-02, prior to fuel load, Watts Bar Unit 2 will install new sump strainers identical to Watts Bar Unit 1. As part of the modification and prior to fuel load, TVA will perform the evaluations to determine that compliance is complete. Due to the Watts Bar Unit 2 construction status and plans to be in compliance prior to fuel load, interim measures are not required.

Attachment 5

NRC BULLETIN 2003-02: LEAKAGE FROM REACTOR PRESSURE VESSEL LOWER HEAD PENETRATIONS AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

To meet the requirements of Bulletin 2003-02, Watts Bar Unit 2 will perform a VT-2 examination of the RPV lower head penetrations during the first refueling outage. At initial startup, Watts Bar Unit 2 will conform to the Corrosion Control Program. Similar to Watts Bar Unit 1, Watts Bar Unit 2 will perform a bare metal visual examination of the 58 RPV lower head penetrations each refueling outage until a change to the ASME Code or a regulatory action justifies a change in frequency.

TVA will perform a baseline inspection prior to fuel load.

Attachment 6

NRC Bulletin 2004-01: Inspection of Alloy 82/182/600 Materials used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors

NRC Requested Information

(1) All subject PWR licensees are requested to provide the following information within 60 days of the date of this bulletin.

(a) A description of the pressurizer penetrations and steam space piping connections at your plant. At a minimum, this description should include materials of construction (e.g., stainless steel piping and/or weld metal, Alloy 600 piping/sleeves, Alloy 82/182 weld metal or buttering, etc.), joint design (e.g., partial penetration welds, full penetration welds, bolted connections, etc.), and, in the case of welded joints, whether or not the weld was stress-relieved prior to being put into service. Additional information relevant with respect to determining the susceptibility of your plant's pressurizer penetrations and steam space piping connections to PWSCC should also be included.

TVA Response: The Watts Bar Unit 2 pressurizer is similar in construction to Watts Bar Unit 1. To provide the information requested requires a comprehensive review of the equipment's original manufacturing records. These records contain proprietary data and are maintained by the original equipment manufacturer (Westinghouse). TVA will provide details of the Unit 2 pressurizer and the penetrations similar to those provided for the Unit 1 pressurizer in Reference 1 by August 15, 2008.

Prior to placing the pressurizer in service, TVA will apply the Material Stress Improvement Process (MSIP) to the Pressurizer Power Operated Relief Valve connections, the safety relief valve connections, the spray line nozzle and surge line nozzle connections.

(b) A description of the inspection program for Alloy 82/182/600 pressurizer penetrations and steam space piping connections that has been implemented at your plant. The description should include when the inspections were performed; the areas, penetrations and steam space piping connections inspected; the extent (percentage) of coverage achieved for each location which was inspected; the inspection methods used; the process used to resolve any inspection findings; the quality of the documentation of the inspections (e.g., written report, video record, photographs); and, the basis for concluding that your plant satisfies applicable regulatory requirements related to the integrity of pressurizer

penetrations and steam space piping connections. If leaking pressurizer penetrations or steam space piping connections were found, indicate what followup NDE was performed to characterize flaws in the leaking penetrations.

TVA Response: The Watts Bar Unit 2 pressurizer has not been placed in service. Prior to placing the pressurizer in service, TVA will apply MSIP to the Pressurizer Power Operated Relief Valve connections, the safety relief valve connections, the spray line nozzle and surge line nozzle connections. The MSIP includes NDE prior to and after completion.

(c) A description of the Alloy 82/182/600 pressurizer penetration and steam space piping connection inspection program that will be implemented at your plant during the next and subsequent refueling outages. The description should include the areas, penetrations and steam space piping connections to be inspected; the extent (percentage) of coverage to be achieved for each location; inspection methods to be used; qualification standards for the inspection methods and personnel; the process used to resolve any inspection indications; the inspection documentation to be generated; and the basis for concluding that your plant will satisfy applicable regulatory requirements related to the structural and leakage integrity of pressurizer penetrations and steam space piping connections. If leaking pressurizer penetrations or steam space piping connections are found, indicate what followup NDE will be performed to characterize flaws in the leaking penetrations. Provide your plans for expansion of the scope of NDE to be performed if circumferential flaws are found in any portion of the leaking pressurizer penetrations or steam space piping connections.

TVA Response: In accordance with NRC Staff recommendations contained in the bulletin, TVA will perform a bare metal visual (BMV) inspection of the upper pressurizer Alloy 600 locations at the first refueling outage. This inspection will be performed utilizing the "in-house" procedure titled "Visual Inspection of Alloy 600/82/182 Pressure Boundary Components." In accordance with plant procedures, personnel performing the inspection will be certified NDE inspectors qualified in the ASME Section XI, VT-2 method. The extent of examination will be 100 percent of each weld circumference and will be documented on written reports which may include photographs or video.

At initial startup, Watts Bar Unit 2 will be under TVA's Corrosion Control Program. This program requires the performance of BMV examinations of Alloy 600/82/182 locations on the upper pressurizer penetrations each refueling outage until further guidance is provided by the Materials Reliability Project.

(d) In light of the information discussed in this bulletin and your understanding of the relevance of recent industry operating experience to your facility, explain why

the inspection program identified in your response to item (1)(c) above is adequate for the purpose of maintaining the integrity of your facility's RCPB and for meeting all applicable regulatory requirements which pertain to your facility.

TVA Response: TVA conducts each inspection with a questioning attitude in accordance with existing industry guidance that includes evaluating and determining the source of any boric acid deposit identified on the upper pressurizer penetrations and the steam space piping. These requirements are incorporated in the visual inspection guidance contained in TVA's Corrosion Control Program and inspection procedures. Implementation of these requirements precludes a through-wall crack remaining undetected for years.

(2) Within 60 days of plant restart following the next inspection of the Alloy 82/182/600 pressurizer penetrations and steam space piping connections, the subject PWR licensees should either:

(a) submit to the NRC a statement indicating that the inspections described in the licensee's response to item (1)(c) of this bulletin were completed and a description of the as-found condition of the pressurizer shell, any findings of relevant indications of through-wall leakage, followup NDE performed to characterize flaws in leaking penetrations or steam space piping connections, a summary of all relevant indications found by NDE, a summary of the disposition of any findings of boric acid, and any corrective actions taken and/or repairs made as a result of the indications found,

or

(b) if the licensee was unable to complete the inspections described in response to item (1)(c) of this bulletin, submit to the NRC a summary of the inspections performed, the extent of the inspections, the methods used, a description of the as-found condition of the pressurizer shell, any findings of relevant indications of through-wall leakage, followup NDE performed to characterize flaws in leaking penetrations or steam space piping connections, a summary of all relevant indications found by NDE, a summary of the disposition of any findings of boric acid, and any corrective actions taken and/or repairs made as a result of the indications found. In addition, supplement the answer which you provided to item (1)(d) above to explain why the inspections that you completed were adequate for the purpose of maintaining the integrity of your facility's RCPB and for meeting all applicable regulatory requirements which pertain to your facility.

TVA Response: TVA plans to submit the required response within 60 days after completion of the first refueling outage.

References:

1. TVA letter dated February 11, 2005, “Sequoyah Nuclear Plant (SQN) Units 1 and 2 and Watts Bar Nuclear Plant (WBN) Unit 1 – Supplemental Response to NRC Bulletin 2004-01, “Inspection of Alloy 82/182/600 Materials used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors” dated May 28, 2004”

Attachment 7

GENERIC LETTER 95-03: CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES

NRC Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes" was issued on April 28, 1995 as a result of then recent nondestructive examination of the steam generator tubing at the Maine Yankee Nuclear Plant which identified a large number of circumferential indications at the top of the tubesheet region, coupled with previously documented inspection results regarding circumferential cracking. The information detailed herein will address the requested actions of Generic Letter 95-03 as they pertain to Westinghouse designed and manufactured steam generators in general, and specifically to WBN Unit 2. WBN Unit 2 has not operated, and as such should not have active tube corrosion phenomena occurring. Additionally, the original WBN Unit 1 SGs, which were identical to the Unit 2 SGs operated successfully for 7 cycles. TVA intends that its actions will be similar to those committed to in the response to GL 95-03 for Unit 1 and this submittal is based on that response.

NRC Requested Action 1:

Evaluate recent operating experience with respect to the detection and sizing of circumferential indications to determine the applicability to their plant.

TVA Response:

TVA's evaluation of operating experience at the time of the submittal of the generic letter for Unit 1 was included in the submittal (Reference 1). That evaluation was used to develop TVA's inspection techniques.

NRC Requested Action 2:

On the basis of the evaluation in Item (a) above, past inspection scope and results, susceptibility to circumferential cracking, threshold of detection, expected or inferred crack growth rates, and other relevant factors, develop a safety assessment justifying continued operation until the next scheduled steam generator tube inspections are performed.

TVA Response:

This request is not applicable to Watts Bar Unit 2. Watts Bar Unit 2 will perform an inspection of 100% of the tubes prior to fuel load.

NRC Requested Action 3:

Develop plans for the next steam generator tube inspections as they pertain to the detection of circumferential cracking. The inspection plans should address, but not be limited to, scope (including sample expansion criteria, if applicable), methods, equipment, and criteria (including personnel training and qualification).

TVA Response:

WBN Unit 2 is constructed with Westinghouse Model D3 steam generators. Unit 2 is under construction and therefore has "0" Effective Full Power Years (EFPY) of operation. Unit 2 will have approximately 1 EFPY of operation at the first scheduled inspection outage.

WBN Areas Susceptible To Circumferential Cracking

The Top-of-Tubesheet (TTS) expansion transition zone is the prevailing tube location in the industry for mainly ODSCC and some PWSCC in Westinghouse Model D plants with full depth hard rolled expansions and I-600MA tubing material. There are a few isolated instances of circumferential indications being reported in Low Row U-Bend locations of I-600MA tubing material for similar plants. WBN Unit 2 has no operating history and therefore denting of tube support plate locations is not a major issue at this time. Preservice inspections have noted a few fabrication related dents at lower TSP locations on the hot leg. WBN Unit 2 has not installed sleeves and is therefore not subject to sleeve related circumferential cracking.

Since WBN Unit 2 has not operated, there have been no primary-to-secondary coolant leaker outages, no pulled tubes, and no gross operating or preoperational chemistry excursions.

Since WBN Unit 2 has not operated, circumferential crack growth rates have not been determined. Industry obtained circumferential crack growth rates for like units will be assumed.

Inservice inspection plans will be similar to WBN Unit 1 Technical Specifications and the latest revision of the EPRI PWR Steam Generator Examination Guidelines. An example of the typical minimum inspection scope, with respect to detecting circumferential cracking at WBN's first refueling outage based on the current version of the Technical Specifications and EPRI guidelines, is as follows:

Base Scope:

20 percent augmented Hot Leg TTS Expansion Zone sample in each steam generator with rotating pancake coil (RPC) or equivalent probe qualified for TTS crack detection.

20 percent augmented Low Row (1 and 2) U-Bend sample in each steam generator with RPC or equivalent probe qualified for U-Bend crack detection.

20 percent augmented dented intersection (greater than or equal to 5 volts by bobbin coil) sample of hot leg TSP 1 and 2 intersections in each steam generator with RPC or equivalent probe qualified for dented intersection crack detection.

The sample is expanded if cracking is detected and, as a minimum, at each of the above areas, the examination expansion requirements of the technical specifications will be fulfilled.

RPC or an equivalent probe qualified to the requirements of EPRI PWR Steam Generator Inspection Guidelines, Appendix H for detection of ODSCC and PWSCC, will be utilized at WBN for detection of circumferential cracking. The use of other supplemental qualified nondestructive examination (NDE) techniques may be used to resolve anomalous/unexpected inservice inspection results.

WBN reviews NDE techniques to:

- Optimize examination methods, minimize noise/interference, and maximize flaw detection.

- Evaluate interfering signals (e.g., lift-off) influence on detection.

- Evaluate examination and analysis procedures to maximize flaw discrimination from unavoidable noise/interference.

- Evaluate examinations for unique unit specific circumstances which necessitate special examination techniques or analysis procedures.

The RPC examination "qualification" requires that a technique demonstrate, at a minimum, a probability of detection (POD) of 80 percent at a 90 percent confidence level for flaws greater than or equal to 60 percent through wall depth on a suitable specimen set as defined by EPRI PWR Steam Generator Examination Guidelines, Appendix H Table S2-2. The actual field performance for qualified techniques is expected to exceed the minimum criteria with the use of conventional RPC for detection of circumferential cracks. This is based on the field data of an industry pulled tube specimen set where the POD is 83 percent at 90 percent confidence level. Only two cracks from the industry pulled tube specimen set were not detected but the maximum depth of those cracks were less than 30 percent through wall.

The EPRI PWR Steam Generator Examination Guidelines provide the direction for developing and applying NDE technology appropriate to manage both existing and emerging damage mechanisms, including circumferential cracking. RPC has been formally qualified per this guideline since 1992 for detection of stress corrosion cracks (irrespective of orientation - axial or circumferential). For circumferentially oriented stress corrosion cracks, field tube pull data indicates that the performance of RPC exceeds the minimum requirements of the EPRI PWR Steam Generator Examination Guidelines, Appendix H for detection. Industry experience indicates that RPC technology applied in adherence with the above protocol have adequately managed circumferential cracking and is based on available tube pull and in-situ burst testing data which indicates structural limits have not been violated.

WBN will utilize qualified MIZ-30 or TC6700 equipment or equally qualified improved equipment as it becomes available. All equipment will be qualified to EPRI PWR Steam Generator Examination Guidelines, Appendix H.

All eddy current Data Analysts will be certified to Eddy Current Level IIA or III.

The first planned steam generator inservice inspection will coincide with the first refueling outage at WBN Unit 2.

References:

1. TVA letter dated June 27, 1995, Watts Bar Nuclear (WBN) – NRC Generic Letter (GL) 95-03 – Circumferential Cracking of Steam Generator Tubes

Attachment 8

NRC GENERIC LETTER 95-05: VOLTAGE-BASED REPAIR CRITERIA FOR WESTINGHOUSESTEAM GENERATOR TUBES AFFECTED BY OUTSIDE DIAMETER STRESS CORROSION CRACKING

No specific written response to this Generic Letter is required. Watts Bar Unit 2 does not currently intend to request a license amendment to implement alternate steam generator tube repair criteria applicable to outside diameter stress corrosion cracking at the tube-to-tube support plate intersections.

Attachment 9

NRC GENERIC LETTER 95-07: PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES

TVA responded to GL 95-07 in references 1 to 4. These responses provided the results of the evaluations performed for pressure locking and thermal binding of safety-related power-operated gate valves and the corrective actions to be taken for those valves identified to be susceptible. NRC closed this issue for Watts Bar Unit 1 in a safety evaluation included in reference 5. TVA intends to use the same approach for Unit 2 as was used for Unit 1. The TVA Watts Bar MOV program includes implementation of GL 95-07 and is described in Maintenance and Modification Department Procedure (MMDP)-5, MOV Program. To support completion of Unit 2, the MOV program will be extended to include Unit 2.

References:

1. TVA letter dated February 13, 1996, Browns Ferry (BFN), Sequoyah (SQN), and Watts Bar (WBN) Nuclear Plants – 180-Day Response to Generic Letter (GL) 95-07 – Pressure Locking and Thermal Binding of Safety Related Power-Operated Gate Valves

2. TVA letter dated March 15, 1996, Browns Ferry (BFN), Sequoyah (SQN), and Watts Bar (WBN) Nuclear Plants – Supplemental Response to Generic Letter (GL) 95-07 – Pressure Locking and Thermal Binding of Safety Related Power-Operated Gate Valves

3. TVA letter dated July 26, 1996, Watts Bar Nuclear Plant (WBN) Unit 1 - Request for Additional Information – Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves

4. TVA letter dated August 2, 1999, Watts Bar Nuclear Plant (WBN) Unit 1 – Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves- Request for Additional Information

5. NRC letter dated September 15, 1999, Watts Bar Unit 1 – Safety Evaluation – Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves

Attachment 10

NRC Generic Letter 96-06: Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions

The GL requested that addressees determine if:

1) containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions;

(2) piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

TVA evaluated both of these conditions for Unit 1 and determined that, with a revision to emergency plan implementation procedures to include a precaution to consider the potential for a waterhammer when restarting essential raw cooling water (ERCW) after a design basis accident, the containment air cooling water systems are not susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions. With procedural draining and assumed valve seat leakage for selected systems, piping systems that penetrate the containment are not susceptible to thermal expansion of fluid which could cause overpressurization of piping. References 1 to 3 include TVA responses to the GL.

For Watts Bar Unit 2, TVA will evaluate both conditions using the same approach as that used on Unit 1.

References:

1. TVA letter dated January 28, 1997, Browns Ferry (BFN), Sequoyah (SQN), and Watts Bar Nuclear Plant (WBN) Response to NRC Generic Letter (GL) 96-06 – Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

2. TVA letter dated December 21, 1998, Sequoyah (SQN) Units 1 and 2, and Watts Bar Nuclear Plant (WBN) Unit 1, Response to NRC Request for Additional Information Regarding Response to NRC Generic Letter (GL) 96-06 – Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

3. TVA letter dated August 31, 1998, Sequoyah (SQN) Units 1 and 2, and Watts Bar Nuclear Plant (WBN) Unit 1, Response to NRC Request for Additional Information Regarding Response to NRC Generic Letter (GL) 96-06 – Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

Attachment 11

NRC GENERIC LETTER 97-04: ASSURANCE OF SUFFICIENT NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL PUMPS

In Generic Letter (GL) 97-04, the NRC staff specifically requested that licensees provide the information outlined below for each of their facilities.

1. Specify the general methodology used to calculate the head loss associated with the ECCS suction strainers.

2. Identify the required NPSH and the available NPSH.

3. Specify whether the current design-basis NPSH analysis differs from the most recent analysis reviewed and approved by the NRC for which a safety evaluation was issued.

4. Specify whether containment overpressure (i.e., containment pressure above the vapor pressure of the sump or suppression pool fluid) was credited in the calculation of available NPSH. Specify the amount of overpressure needed and the minimum overpressure available.

5. When containment overpressure is credited in the calculation of available NPSH, confirm that an appropriate containment pressure analysis was done to establish the minimum containment pressure.

Watts Bar Unit 2 will utilize the same methodology as Watts Bar Unit 1. In response to GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors", Watts Bar Unit 2 will install new larger suction strainers. Part of the design change will be a revision to the NPSH calculations. NRC reviewed the Watts Bar Unit 1 NPSH calculations as part of an audit of GL 2004-02 activities. NRC concluded that TVA's use of a sump pool temperature of 190ºF and taking no credit for containment overpressure was acceptable.

Attachment 12

GENERIC LETTER (GL) 97-05: STEAM GENERATOR TUBE INSPECTION TECHNIQUES

NRC Required Action 1:

Inform NRC if it is the licensee's practice to leave steam generator tubes with indications in service based on sizing.

TVA's Response

It will be the practice at WBN Unit 2, consistent with the Unit 1 approach, to leave certain steam generator tubes with indications in-service based on sizing if the indications are less than the 40 percent of the technical specification plugging limit. However, WBN does not leave crack-like indications in-service.

NRC Required Action 2:

If the response to item (1) is affirmative, those licensees should submit a written report that includes, for each type of indication, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the techniques used.

TVA Response:

TVA will employ the same approach as was used on the original Unit 1 steam generators using the Electric Power Research Institute (EPRI) PWR Steam Generator Examination Guidelines, Appendix H, "Performance Demonstration for Eddy Current Examination," Revision 6.

Attachment 13

NRC GENERIC LETTER (GL) 97-06: DEGRADATION OF STEAM GENERATOR (SG) INTERNALS

GL 97-06, "Degradation of Steam Generator Internals," was issued to: alert addressees to findings of damage to SG internals, namely, tube support plates (TSPs) and tube bundle wrappers; emphasize the importance of performing comprehensive examinations of SG internals to ensure SG tube structural integrity is maintained in accordance with the requirements of Appendix B to 10 CFR Part 50; and require all addressees to submit information that will enable the NRC staff to verify whether addressees' SG internals comply with and conform to the current licensing bases for their respective facilities.

Prior to the issue of the GL, the Westinghouse Owners Group (WOG), Electric Power Research Institute (EPRI), and Nuclear Energy Institute (NEI) developed an action plan to assess susceptibility to secondary side degradation, which included a requirement to understand the causal factors involved in the degradation first experienced in the Electricite de France (EDF) units. This information is captured in EPRI report GC-109558, "Steam Generator Internals Degradation: Modes of Degradation Detected in EDF Units." This report was submitted to the NRC via NEI letter dated December 19, 1997.

The WOG report on this subject for Series 51 SGs (WCAP-15002, Revision 1) determined limited potential susceptibility and concluded that the number of plants that were inspected and the inspection results demonstrate that the causal factors identified for damage do not jeopardize the continued operability of Westinghouse Series 51 SGs. Eddy current inspection of the tubes would detect any detrimental effects on the tubing due to wear caused by TSP ligament degradation, wear due to loose parts, and wear due to secondary side flow distribution changes. Foreign object search and retrieval (FOSAR) efforts are conducted to discover loose parts.

Below are the responses to the NRC requests in GL 97-06.

NRC Requested Action 1:

A discussion of any program in place to detect degradation of SG internals and descriptive inspection plans, including the inspection scope, frequency, methods, and equipment.

TVA Response:

As discussed in WCAP-15002, Revision 1, surveys were sent to WOG utilities requesting the results of SGs secondary side inspections and relevant tube inspections for TSP conditions. Completed surveys were received for 37 of 49 plants. For the Model D, E and F SGs, responses were received for 12 plants. Eleven of these plants responded as having inspected or reviewed inspection data for TSP ligament indications and 8 having performed SG secondary side entries that give confidence of not having wrapper drop. TSP ligament indications were not found in either SGs with carbon steel or with stainless steel support plates.

The modes of degradation detected include many cases of flow-assisted corrosion or erosion-corrosion of upper internals components and of premature cracking of shell welds that results from either surface fatigue or from corrosion cracking that is associated with surface conditions such as pitting. For the most part, however, the surveys did not report detection of several modes of degradation experienced in the damaged units. There was no evidence of post-chemical cleaning inspections discovering any significant material losses. There was no evidence of any wrapper having dropped. There was no evidence of TSP ligament cracking or thinning that was progressive and continuing. TSP ligament cracking or missing pieces of ligaments have been observed, but only in units with carbon steel support plates with drilled round tube holes and flow holes. These conditions are generally traceable to initial inspections and are not progressing based on sequential inspection data. Many of the conditions are probably related to original TSP drilling alignment. Cases of TSPs with indications have been identified which have been linked to welded patch plates.

Plants with significant hour-glassing of the TSPs as a result of the denting process have exhibited ligament cracking throughout the thickness of the support plate between the flow holes in the plate or the flow holes in the tube lane. If denting remained uncontrolled, as subsequent support plate corrosion occurs, the potential exists for fragments of the support plate material to become completely free of the main TSP structure. However, these plate segments generally remain locked in place because of the in-plane forces that give rise to denting, as well as the deformation that contains the individual pieces. Operating plants with active denting are under periodic monitoring by the utility and have long-standing criteria and review by the NRC.

Based on the above history and inspection performed on the original WBN Unit 1 SGs, the following inspection plan will be implemented for Unit 2. Except where noted, these inspections will be performed during each refueling outage.

Tube Support Plate Erosion-Corrosion and Cracking:

1. Because the TSPs in WBN SGs are made of carbon steel, a pre-service baseline will be performed employing a bobbin inspection technique. A bobbin coil inspection technique will be used during each outage. The technique to be employed is defined in the EPRI Report, SG-96-05-003, "Investigation of Applicability of Eddy Current to the Detection of Potentially Degraded Support

Structures," dated May 1996. If indications are found, the history is reviewed to establish if this is an active degradation mechanism and an evaluation is performed to determine structural significance.

2. In-service inspection will be conducted in accordance with Revision 6 of the EPRI PWR SG Examination Guidelines.

The critical area for mechanical or thermally induced support plate cracking will tentatively be defined as a region three tubes deep around the periphery and a region two rows deep around the patch plate joint in each support plate. The critical area for ligament erosion/corrosion is the entire bundle.

During eddy current inspections, the bobbin coil data acquired during examination is evaluated for indications of TSP degradation.

Wrapper Drop:

Design of Model D plants preclude wrapper drop.

1. A determination will be made that the sludge lance equipment can be inserted into the sludge lance ports without interference. WBN will perform sludge lancing each outage.

2. A visual inspection will be conducted on the lower wrapper support blocks, if interference with the sludge lance equipment is detected.

Wrapper Cracking:

No inspection is recommended unless evidence of wrapper misposition or tube damage in the periphery of the first TSP is detected. A visual inspection will be conducted on the lower wrapper support blocks, if degradation is detected.

Upper Package:

Upper internals visual inspections will be performed on a frequency that ensures each SG is inspected every six years. This inspection is included in site maintenance procedures. FOSAR will be performed each outage.

Transition Cone Girth Weld:

Inspections will performed in accordance with the SG shell, Section XI in-service inspection requirements. Visual inspections are required during SG upper internals inspections.

NRC Requested Action 2:

If the addressee currently has no program in place to detect degradation of SG internals, include a discussion and justification of the plans and schedule for establishing such a program, or why no program is needed.

TVA Response:

Item 2 of the GL does not apply to WBN.

Reference

1. WCAP-15002, Revision 1, "Evaluation of EDF Steam Generator Internals Degradation - Impact of Causal Factors on Westinghouse Series 51 Steam Generators"

Attachment 14

NRC GENERIC LETTER (GL) 98-02: Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition

The GL requested that addressees (1) perform an assessment of whether their emergency core cooling systems include certain design features, which can render the systems susceptible to common-cause failure as a result of events similar to the Wolf Creek reactor coolant system (RCS) drain-down event; and if this susceptibility is found, (2) prepare, with consideration of plant-specific design attributes, a description of the features of their Appendix B quality assurance program that provide assurance that the safety-related functions of the residual heat removal (RHR) system and emergency core coolant system (ECCS) will not be adversely affected by activities conducted at hot shutdown.

The TVA review of relevant flow paths did not identify specific vulnerabilities which could reasonably be expected to result in a significant flow of hot RCS water to the refueling water storage tank (RWST)/ECCS header, and no corrective actions were identified as a result of this review. Reference 1 provides the information requested by NRC for Watts Bar Unit 1 and also indicates that a report summarizing 10CFR50 Appendix B controls that will act to prevent, or assist in the mitigation of, such an event had been prepared and was retained for NRC inspection. For Watts Bar Unit 2, TVA will perform a similar review and document the results.

References:

1. TVA letter dated November 24, 1998, Sequoyah Nuclear Plant (SQN), and Watts Bar Nuclear Plant, 180-Day Response to Generic Letter 98-02, Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition

Attachment 15

NRC GENERIC LETTER 98-04: POTENTIAL FOR DEGRADATION OF THE EMERGENCY CORE COOLING SYSTEM AND THE CONTAINMENT SPRAY SYSTEM AFTER A LOSS-OF-COOLANT ACCIDENT BECAUSE OF CONSTRUCTION AND PROTECTIVE COATING DEFICIENCIES AND FOREIGN MATERIAL IN CONTAINMENT

TVA responded to Generic Letter (GL) 98-04 in Reference 1. The responses provided in Enclosure 3 of Reference 1 are also applicable to Watts Bar Unit 2 with the exception of the amount of unqualified coatings.

The amount of Watts Bar Unit 2 unqualified coatings will be documented as part of the strainer replacement associated with GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors". As part of the modification, TVA will perform the necessary containment walkdowns, debris generation study, debris transport analysis, chemical effects and downstream effects analysis. These analyses will verify that the Watts Bar Unit 1 analyses bound Watts Bar Unit 2. TVA will also inspect and repair service level I coatings. The programmatic controls that ensure potential sources of debris introduced into containment will be assessed for potential adverse effects will be put in place prior to fuel load.

References:

1. TVA letter to NRC dated November 10, 1998," Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN) and Watts Bar Nuclear Plant (WBN), 120-day Response Generic Letter (GL) 98-04, "Potential for Degradation of the Emergency Core Cooling System (ECCS) and the Containment Spray System (CSS) after a Loss-of-Cooling Accident (LOCA) Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," Dated July 14, 1998"

Attachment 16

NRC GENERIC LETTER 2003-01: CONTROL ROOM HABITABILITY

TVA responded to Generic Letter (GL) 2003-01 in Reference 1. The Watts Bar Unit 2 Control Room is part of the Watts Bar Unit 1 Main Control Room Habitability Zone (MCRHZ). The MCRHZ is periodically tested per the Watts Bar Unit 1 Technical Specification requirements. The responses to the NRC questions in Reference 1 are applicable to Watts Bar Unit 2.

Watts Bar Unit 2 modifications that penetrate the MCRHZ boundary will be performed in a manner to maintain the operability of the boundary to support Unit 1 operation.

TVA will incorporate the technical specification surveillance requirement from Technical Specification Task Force (TSTF) – 448 into the Watts Bar Unit 2 Technical Specification submittal.

References:

1. TVA letter dated August 4, 2004," Watts Bar Nuclear Plant (WBN) Unit 1 – Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2003-01: Control Room Habitability – Final Response (TAC MB 9872)"

Attachment 17

NRC GENERIC LETTER 2004-01: REQUIREMENTS FOR STEAM GENERATOR TUBE INSPECTIONS

NRC Request No. 1

Within 60 days of the date of this generic letter, addressees are requested to provide the following information to the NRC:

Addressees should provide a description of the SG tube inspections performed at their plant during the last inspection. In addition, if they are not using SG tube inspection methods whose capabilities are consistent with the NRC's position, addressees should provide an assessment of how the tube inspections performed at their plant meet the inspection requirements of the TS in conjunction with Criteria IX and XI of 10 CFR Part 50, Appendix B, and corrective action taken in accordance with Appendix B, Criterion XVI. This assessment should also address whether the tube inspection practices are capable of detecting flaws of any type that may potentially be present along the length of the tube required to be inspected and that may exceed the applicable tube repair criteria.

TVA Response:

WBN Unit 2 SGs are Westinghouse Model D3 with Alloy 600 low temperature mill annealed ¾ inch Outside Diameter, 0.043 inch wall tubing with full depth hard-rolled tubesheet. They are the same design as the original Unit 1 SGs and have never been in service.

An informational inspection of the Unit 2 SG tubes was performed in February 2007 to determine if lay-up conditions had contributed to tube and/or support plant degradation, and to assess their general condition. This included all four SGs. A bobbin probe was used on a 25% systematic sample from each SG to assess their condition. The results indicated that the tubes are in good condition and show no detrimental effects from lay-up. There are some tube ends damaged at the tubesheet. TVA will perform a complete 100% SG inspection prior to fuel load.

The WBN Unit 2 SG tube inspection method will be consistent with NRC's position that "licensees are required under existing requirements (Technical Specifications in conjunction with 10 CFR Part 50, Appendix B) to employ inspection techniques capable of detecting all flaw types which may be present at locations which are required to be inspected pursuant to the TS." Therefore, the remainder of the requested information is not applicable to WBN Unit 1.

NRC Request # 2

If addressees conclude that full compliance with the TS in conjunction with Criteria IX, XI and XVI of 10 CFR Part 50, Appendix B, requires corrective actions, they should discuss their proposed corrective actions (e.g., changing inspection practices consistent with the NRC's position or submitting a TS amendment request with the associated safety basis for limiting the inspections) to achieve full compliance. If addressees choose to change their TS, the staff has included in the attachment suggested changes to the TS definitions for a tube inspection and for plugging limits to show what may be acceptable to the staff in cases where the tubes are expanded for the full depth of the tubesheet and where the extent of the inspection in the tubesheet region is limited.

TVA Response

WBN Unit 2 SG tube inspection practice will be consistent with NRC's position. Therefore, the remainder of the requested information is not applicable.

NRC Request # 3

For plants where SG tube inspections have not been or are not being performed consistent with the NRC's position on the requirements in the TS in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B, the licensee should submit a safety assessment (i.e., a justification for continued operation based on maintaining tube structural and leakage integrity) that addresses any differences between the licensee's inspection practices and those called for by the NRC's position. Safety assessments should be submitted for all areas of the tube required to be inspected by the TS where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used, and should include the basis for not employing such inspection techniques. The assessment should include an evaluation of (1) whether the inspection practices rely on an acceptance standard (e.g., cracks located at least a minimum distance of x below the top of the tube sheet, even if these cracks cause complete severance of the tube) which is different from the TS acceptance standards (i.e., the tube plugging limits or repair criteria), and (2) whether the safety assessment constitutes a change to the "method of evaluation" (as defined in 10 CFR 50.59) for establishing the structural and leakage integrity of the joint. If the safety assessment constitutes a change to the method of evaluation under 10 CFR

50.59, the licensee should determine whether a license amendment is necessary pursuant to that regulation.

TVA Response:

The WBN Unit 1 inspection practice is consistent with the NRC position. Therefore this item is not applicable and a response is not required.

Attachment 18

NRC GENERIC LETTER 2004-02: POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS

TVA responded to Generic Letter (GL) 2004-02 in References 1 to 3. The responses provided for Watts Bar Unit 1 are applicable to Watts Bar Unit 2. Prior to fuel load, Watts Bar Unit 2 will install new sump strainers identical to Watts Bar Unit 1. As part of the modification, TVA will perform the necessary containment walkdowns and analysis (debris generation study, debris transport analysis, chemical effects and downstream effects analysis) for Watts Bar Unit 2. TVA will inspect and repair service level I coatings and limit fibrous insulation to the extent practicable. The programmatic controls that ensure potential sources of debris introduced into containment are assessed for potential adverse effects will be put in place prior to fuel load.

The principal differences between Watts Bar Unit 1 and 2 that are recognized at this time are:

- Unit 2 will limit the use of 3M fire barriers and min-K insulation materials in the lower containment to the extent practicable, and
- Watts Bar Unit 2 steam generators are coated.

TVA will provide a supplemental response for Watts Bar Unit 2 similar to Reference 4 to provide the unit specific information requested in Reference 5. This information will be provided by April 1, 2009.

References:

1. TVA letter dated March 7, 2005,"Sequoyah Nuclear Plant (SQN) and Watts Bar Nuclear Plant (WBN) – Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors (PWR) – 90-Day Response
2. TVA letter dated July 21, 2005,"Sequoyah Nuclear Plant (SQN) and Watts Bar Nuclear Plant (WBN) – Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors (PWR) – Request for Additional Information (RAI) (TAC NOS. MC4717, MC4718 and MC4730)
3. TVA letter dated September 1, 2005," Watts Bar Nuclear Plant (WBN) – Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized- Water Reactors (PWR) – Second Response (TAC NO. MC4730).

4. TVA letter dated April 11, 2006," Watts Bar Nuclear Plant (WBN) – Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized- Water Reactors (PWR) – Response to Request for Additional Information (TAC NO. MC4730)
5. NRC letter dated February 10, 2006," Watts Bar Nuclear Plant, Unit 1, Request for Additional Information Re: Response to Generic Letter 2004-02: "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized- Water Reactors" (TAC NO. MC4730)

Attachment 19

NRC GENERIC LETTER 2006-01: STEAM GENERATOR TUBE INTEGRITY AND ASSOCIATED TECHNICAL SPECIFICATIONS

TVA submitted a request to NRC for WBN Unit 1 to modify the Steam Generator (SG) portion of the Technical Specifications (TS) consistent with the TS Task Force (TSTF) Standard TS Traveler, TSTF-449, Steam Generator Tube Integrity, Revision 4. TVA will include the TSTF in the WBN Unit 2 Technical Specifications submittal.

References:

1. TVA letter dated February 21, 2006, "Sequoyah Nuclear Plant (SQN) Units 1 and 2 and Watts Bar Nuclear Plant (WBN) Unit 1 – Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2006-01: Steam Generator Tube Integrity and Associated Technical Specifications – Response"

Attachment 20

NRC GENERIC LETTER 2006-02: GRID RELIABILITY AND THE IMPACT ON PLANT RISK AND THE OPERABILITY OF OFFSITE POWER

TVA responded to Generic Letter (GL) 2006-02 in Reference 1. TVA responded to a request for additional information in Reference 2. The responses to NRC question 1-8 are generic to all TVAN Nuclear Units including Watts Bar Unit 2. The offsite power and interconnections are common to both Watts Bar Unit 1 and Unit 2. In order to demonstrate compliance with GDC 17, the two-unit baseline electrical calculations and revisions to the implementing procedures are required prior to fuel load. This action was previously committed for Unit 2 in Reference 3.

With respect to NRC question 9, Watts Bar Unit 2 will be in compliance with the applicable regulations prior to fuel load.

References:

1. TVA letter dated April 3, 2006, "Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3, Sequoyah Nuclear Plant (SQN) Units 1 and 2 and Watts Bar Nuclear Plant (WBN) Unit 1 – Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2006-02: Grid Reliability and the Impact of Plant Risk and the Operability of Offsite Power – Response"

2. TVA letter dated January 31, 2007, "Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3, Sequoyah Nuclear Plant (SQN) Units 1 and 2 and Watts Bar Nuclear Plant (WBN) Unit 1 – Request for Additional Information Regarding Resolution of Generic Letter 2006-02: Grid Reliability and the Impact of Plant Risk and the Operability of Offsite Power (TAC Nos. MD0947 through MD1050"

3. TVA letter dated October 9, 1990, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 – Lack of Adequate Calculations to Document Electriacl System Design Basis – WBRD-50-390/86-17 and WBRD-50-391/86-13 – Revised Final Report"

Attachment 21

NRC GENERIC LETTER 2006-03: POTENTIALLY NONCONFORMING HEMYC AND MT FIRE BARRIER CONFIGURATIONS

TVA responded to Generic Letter (GL) 2006-03 in Reference 1. The responses to the NRC questions are generic to all TVA Nuclear Units and also apply to Watts Bar Unit 2. In summary, TVA does not rely on Hemyc or MT materials to protect electrical and instrumentation cables or equipment that provide safe shutdown capability during a postulated fire.

TVA relies on Thermo-Lag fire barrier material to protect fire safe shutdown circuits. Thermo-Lag has undergone extensive testing by both the industry and TVA. These tests were developed consistent with the guidance contained in the applicable codes, standards and regulatory guidance. Configurations installed at TVA facilities are in accordance with the tested configurations or have been evaluated by persons knowledgeable in fire barrier design and installation. The results of both the testing and engineering evaluations have been documented consistent with accepted engineering and industry standards. These configurations, both those specifically tested and unique configurations, are documented in facility design basis documentation that are controlled and maintained in accordance with TVA's Design Control and Quality Assurance Programs. The Fire Protection Corrective Action Program will ensure Watts Bar Unit 2 conforms with NRC requirements and applicable guidelines prior to fuel load.

References:

1. TVA letter dated June 7, 2006, “Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3, Sequoyah Nuclear Plant (SQN) Units 1 and 2 and Watts Bar Nuclear Plant (WBN) Unit 1 – Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2006-03: Potentially Nonconforming Hemyc and MT Fire Barrier Configurations – 60 Day Response”

Attachment 22

NRC Generic Letter 2007-01: Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients.

TVA responded to Generic Letter 2007-01 in Reference 1. The response is generic to all TVAN Nuclear Units. The response indicates that there were 20 Watts Bar cables in the test program. This included the following cables:

8 Essential Raw Cooling Water pump cables which are common to both units
4 Diesel Generator cables which are common to both units
4 Unit 1 Reactor Coolant Pump (RCP) cables
4 Unit 1 Condenser Circulating Water (CCW) pump cables

The Unit 2 RCP cables will not be routed in an underground duct bank and are therefore not applicable.

The 4 Unit 2 CCW pump cables will in addition have to be tested for Unit 2 startup. Therefore the total number of cables identified should be changed from 20 to 24. Watts Bar Unit 2 will complete the testing of these 4 additional cables before fuel load.

References:

1. TVA letter dated May 4, 2007," Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3, Sequoyah Nuclear Plant (SQN) Units 1 and 2, and Watts Bar Nuclear Plant (WBN) – Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2007-01: Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients – 90 Day Response"

Attachment 23

Commitment Summary

1. Bulletin 96-01 – Control Rod Insertion Problems

   The current Emergency Operating Instruction ES-0.1 has the Reactor Operator ensure all control rods are fully inserted as indicated by the rod position indication system. This procedure initiates boration if two or more control rods are not fully inserted. A similar procedure will be issued for Unit 2 prior to startup.

   TVA will provide a core map of rodded fuel assemblies indicating fuel type (materials, grids, spacers, guide tube inner diameter) and projected end of cycle burnup of each rodded assembly for the initial fuel cycle. This information will be provided six months prior to fuel load.

2. Bulletin 01-01 – Cracking of RPV Head Penetration Nozzles

   To meet the requirements of Bulletins 2001-01, 2002-01 and 2002-02, Watts Bar Unit 2 will perform the first inspections meeting the requirements of paragraphs IV.C(5)(a) and IV.C(5)(b) of NRC Order EA-03-009 at the first refueling outage.

   TVA will perform a baseline inspection prior to fuel load.

3. Bulletin 02-01 - RPV Head Degradation and Reactor Coolant Pressure Boundary Integrity

   See item 2

4. Bulletin 02-02 – RPV Head and Vessel Head Penetration Nozzle Inspection Program

   See item 2

5. Bulletin 03-02 – Leakage from RPV Lower Head Penetrations & Reactor Coolant Pressure Boundary Integrity

   Watts Bar Unit 2 will perform a VT-2 examination of the RPV lower head penetrations during the first refueling outage.

   TVA will perform a baseline inspection prior to fuel load.

6. Bulletin 04-01 – Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at PWRs

   TVA will provide details of the Unit 2 pressurizer and the penetrations similar to those provided for the Unit 1 pressurizer in Reference 1 by August 15, 2008.

   Prior to placing the pressurizer in service, TVA will apply the Material Stress Improvement Process (MSIP) to the Pressurizer Power Operated Relief Valve connections, the safety relief valve connections, the spray line nozzle and surge line nozzle connections.

   TVA will perform a bare metal visual (BMV) inspection of the upper pressurizer Alloy 600 locations at the first refueling outage.

   TVA plans to submit the required response within 60 days after completion of the first refueling outage.

7. Generic Letter 95-03 – Circumferential Cracking of Steam Generator (SG) Tubes

   See item 15

8. Generic Letter 95-07 – Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves

   To support completion of Unit 2, the MOV program will be extended to include Unit 2.

9. Generic Letter 96-06 – Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

   TVA will evaluate both conditions using the same approach as that used on Unit 1.

10. Generic Letter 97-04 – Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps

    See item 16

11. Generic Letter 97-06 – Degradation of Steam General Internals

The inspection plan discussed in Attachment 13 will be implemented for Unit 2. These inspections will be performed during each refueling outage.

12. Generic Letter 98-02 – Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition

    TVA will perform a similar review on Unit 2 and document the results.

13. Generic Letter 98-04 – Potential for Degradation of the ECCS and the Containment Spray System After a LOCA Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment

    See item 16

14. Generic Letter 03-01 – Control Room Habitability

    TVA will incorporate the technical specification surveillance requirement from Technical Specification Task Force (TSTF) – 448 into the Watts Bar Unit 2 Technical Specification submittal.

15. Generic Letter 04-01 – Requirements for SG Tube Inspection

    TVA will perform a complete 100% SG inspection prior to fuel load.

16. Generic Letter 04-02 – Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at PWRs

    Prior to fuel load, Watts Bar Unit 2 will install new sump strainers and perform other modification related actions identical to Watts Bar Unit 1.

    TVA will provide a supplemental response for Watts Bar Unit 2 similar to that provided for Unit 1 to provide the unit specific information requested by NRC, by April 1, 2009.

17. Generic Letter 06-01 – SG Tube Integrity and Associated Technical Specifications

    TVA will include TSTF-449 in the WBN Unit 2 Technical Specifications submittal.

18. Generic Letter 06-03 – Potentially Nonconforming Hemyc and MT Fire Barrier Configurations

The Fire Protection Corrective Action Program will ensure Watts Bar Unit 2 conforms with NRC requirements and applicable guidelines prior to fuel load. The fire barrier configurations are documented in facility design basis documentation that are controlled and maintained in accordance with TVA's Design Control and Quality Assurance Programs.

19. Generic Letter 07-01 – Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients

    Watts Bar Unit 2 will complete the testing of these 4 additional cables before fuel load.